ACCESSION #: 9602280354

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Palo Verde Unit 2 PAGE: 1 OF 7

DOCKET NUMBER: 05000529

TITLE: Improper Venting Of Condensate Pump Results In Loss Of

Feedwater Flow and Reactor Trip

EVENT DATE: 01/21/96 LER #: 96-001-00 REPORT DATE: 02/20/96

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Burton A. Grabo, Section TELEPHONE: (602) 393-6492

Leader, Nuclear Regulatory Affairs

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 21, 1996, at approximately 1130 MST, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when a reactor trip occurred when Steam Generator Number 1 (SG-1) water level reached the Reactor Protection System trip setpoint for low SG water level following the degradation of main feedwater (FW) flow. Immediately following the reactor trip an Engineered Safety Feature Actuation System (ESFAS) actuation of the Auxiliary Feedwater Actuation System (AFAS) was received on SG-1 and SG-2 due to low SG water levels. The loss of

FW flow was due to the improper filling and venting of the "C" condensate pump being restored to service after maintenance. When the condensate pump suction valve was partially opened, to allow filling of the suction piping, the two running condensate pumps became air bound. This resulted in the main FW pumps tripping on low suction pressure and the reactor tripping on SG low water level. As corrective action, filling and venting of condensate pumps has been proceduralized.

There have been no previous similar events reported pursuant to 10CFR50.73.

TEXT PAGE 2 OF 7

1. REPORTING REQUIREMENT:

This LER 529/96-001-00 is being written to report an event that resulted in the automatic actuation of an Engineered Safety Feature (ESF)(JE), including the Reactor Protection System (RPS)(JC) as specified in 10 CFR 50.73 (a) (2) (iv).

Specifically, at approximately 1130 MST on January 21, 1996, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when a reactor trip occurred when Steam Generator Number 1 (SG-1)(AB) water level reached the RPS trip setpoint for low SG water level following the degradation of main feedwater (MFW)(SJ) flow. Immediately following the reactor trip an Engineered Safety Feature Actuation System (ESFAS)(JE) actuation of the Auxiliary Feedwater Actuation System (AFAS 1 and AFAS 2) (JE, BA) was received on SG-1 and SG-2 low water levels.

Required plant equipment and safety systems responded to the event as designed. No other safety actuations occurred and none were required. By approximately 1230 MST on January 21, 1996, the plant was stabilized in Mode 3 (HOT STANDBY).

2. EVENT DESCRIPTION:

On January 21, 1996, at approximately 0930 hours MST, Unit 2 was in Mode 1 (POWER OPERATION) at approximately 100 percent power when Operations personnel (utility, licensed) held a pre-job briefing to return the "C" condensate (SD) pump to service following maintenance activities.

At approximately 1100 hours, after filling the condensate pump minimum recirculation and overboard line, the pump casing and discharge vent valves were opened and throttled open (respectively). Control room personnel noticed a decrease in condenser vacuum and notified the Reactor operator performing the condensate pump restoration of the condition. Condenser vacuum was stabilized by throttling closed the discharge vent valve. The discharge vent valve was gradually reopened to continue the restoration process.

At approximately 1125 hours, the control room was informed that the suction valve for the "C" condensate pump was about to be opened so that condenser hotwell level and vacuum could be monitored. When the suction

valve was partially opened a change in noise level was recognized and the suction valve was immediately closed. The operator attempted to inform the control room of the condition and was informed that the reactor had tripped.

At approximately 11:29:43 hours, main feedwater pump "B" (MFP) tripped on low suction pressure. A reactor power cutback actuated, when MFP "B" tripped, which decreased power to approximately sixty percent. MFP "A" tripped approximately seven seconds later on low suction pressure.

At approximately 11:30:15 hours, a reactor trip occurred on low water level in SG-1. SG levels continued to decrease until an actuation of AFAS 1 and 2 occurred at 11:30:46 hours. Required plant equipment and safety systems responded to the event as designed. No other safety system actuations occurred and none were required. The reactor trip was classified as uncomplicated by the Shift Supervisor (utility, licensed) at 1151 hours.

At approximately 1230 hours, after the EOPs were exited, operation personnel (utility, licensed) performed surveillance test procedure 72ST-9RX09, "Shutdown Margin".

Current administrative limits require shutdown margin (SDM) to be greater than or equal to 6.5 percent delta k/k for Mode 3 (HOT STANDBY) greater than 500 degrees Fahrenheit (F). With SDM less than 6.5 percent delta k/k, immediately initiate and continue boration at greater than or equal to twenty-six gallons per minute (gpm) to the reactor coolant system of a solution containing greater than or equal to 4000 part per million (ppm) boron or equivalent until the required SHUTDOWN MARGIN is restored.

The Control Room Supervisor (CRS) was informed that 72ST-9RX09 required a boration because SDM could not be verified. The CRS directed that an additional 400 gallons of borated water be added to the reactor coolant system (RCS)(AB). While waiting for RCS chemistry sample results, the Shift Technical Advisor (STA) (utility, nonlicensed) identified that procedure 72ST-9RX09 required a continuous boration until SDM could be formally verified. At approximately 1405 hours, a boration was initiated and SDM was verified to be adequate at 1423 hours.

3. ASSESSMENT OF THE SAFETY CONSEQUENCES AND THE IMPLICATIONS OF THIS

EVENT:

Prior to the reactor trip signal resulting from low steam generator water level in SG-1, pressurizer pressure peaked at 2275 psia. The peak pressure criteria of 110 percent of design (2750 psia) was never challenged during this reactor coolant system (RCS)(AB) transient. The steam generator pressure peaked at 1190 psia and was well below the pressure criteria of 110 percent of secondary design pressure (1398 psia).

This Unit 2 reactor trip was classified as a reactor trip on loss of feedwater (LFW) in the Anticipated Operational Occurrence (AOO) category, which is a moderate frequency event. This event did not challenge the shutdown margin criteria. All CEAs inserted as designed. Equipment and systems assumed in the Safety Analysis were functional. Plant response was normal for the situation that occurred.

Scenarios defined in the USFAR Chapter 15 and design assumptions of the RPS will remain bounding for this Unit 2 reactor trip.

Scenarios defined in the UFSAR Chapter 6, concerning Loss of Coolant Accidents were not implicated by this transient. The transient did not cause any violation of the Specified Acceptable Fuel Design Limits (SAFDLs).

Subsequent analyses have demonstrated there was adequate shutdown margin and Technical Specifications were met following the reactor trip even though the SDM was not satisfied per 72ST-9RX09 due to the procedure's built-in conservative uncertainties.

This event did not result in any challenges to the fission product barriers or result in any releases of radioactive materials.

Therefore, there were no adverse safety consequences or implications as a result of this event. This event did not adversely affect the safe operation of the plant or the health and safety of the public.

4. CAUSE OF THE EVENT:

An independent investigation of this event has been conducted in accordance with the APS Corrective Action Program. Based on the results of the independent investigation, the cause of the reactor trip is that condensate pump "C" was improperly filled and vented while restoring the pump for service (SALP Cause Code A: Personnel Error).

At the time of the event, filling and venting a condensate pump was not proceduralized. The operating crew restoring the condensate pump to service developed an action plan and held a pre-job briefing for the pump restoration. However, the action plan was inadequate to ensure proper venting. No unusual characteristics of the work location (e.g., noise, heat , poor lighting) directly contributed to this event.

5. STRUCTURE, SYSTEM, OR COMPONENT INFORMATION:

No structures, systems, or components were inoperable at the start of the event which contributed to this event. No failures of components with multiple functions were involved. No failures that rendered a train of safety system inoperable were involved. All safety system actuations that were required actuated as designed.

6. CORRECTIVE ACTIONS TO PREVENT RECURRENCE:

On February 1, 1996 procedure 4XOP-XZZ14, "Feedwater and Condensate", was revised to include instructions for filling and venting a condensate pump.

On January 22, 1996, a night order was issued to all three units to reiterate that if SDM is not satisfied during the performance of

7XST-9RX09 "Shutdown Margin", a boration will be started and continued until a valid chemistry sample verifies that the minimum boron concentration has been reached and the surveillance procedure (RX09) is completed satisfactorily.

Actions required from the above investigation will be tracked by APS' Commitment Action Tracking System (CATS)

7. PREVIOUS SIMILAR EVENTS:

There have been six reactor trips due to SG low level reported pursuant to 10CFR50.73 in the last three years (LERs 529/92-001, 529/93-004, 530/94001, 530/94-005, 528/95-008, and 529/95-005). The loss of main feedwater reported in these previous LERs were initiated by means other than improper venting. Corrective actions taken for the previous events would not have prevented this event.

8. ADDITIONAL INFORMATION:

Soon after the reactor power cutback, MFP "A" tripped and within four seconds suction pressure increased until it reached equilibrium with the MFP discharge pressure. Maintenance Engineering personnel (utility, nonlicensed) performed a walk-down of the Turbine Building and determined that there was no damage to the pipe support systems, however a leak was found on MFP "B" suction strainer.

A review of the parameters associated with the over-pressurization of the MFP suction piping has revealed that the discharge check valve of MFP "A" may have leaked past its seat enough to cause the suction pressure of both pumps to increase to pump discharge pressure in four seconds.

Engineering personnel (utility, nonlicensed) have evaluated the over-pressurization and has determined that this event is similar to the 1989 over-pressurization event. The difference is that the 1996 event caused an internal pressure of 1020 psig for seven minutes at operating temperature of 350 degrees where as the 1989 event caused a pressure of 1580 psig for one minute at operating temperature of 100 degrees.

The resultant stresses for the 1996 event are lower overall. Cyclic fatigue is not a concern since the two cycles caused by the 1989 and 1996 events have remained in the elastic range where the allowable cycles are very large. Therefore, the existing stress margins are not decreased and the existing 1989 evaluation is still bounding.

On January 23, 1996, MFP "A" discharge check valve was inspected. The disc was able to be physically moved through the full range of motion and no abnormal resistance to movement was identified. In general the internal surfaces had no indications of abnormal wear or physical damage.

The top of one hinge pin was observed to be in full contact with the integral hinge pin stop on the valve body. Although there was some gap between the hinge pin head (underside) it was determined to be insufficient to allow the disc to move laterally if required to

center the valve seat.

Mechanical Maintenance (utility, nonlicensed) reworked the valve to

remove metal from the west side body stop to obtain a 0.125 inch

gap. After

rework a 0.010 to 0.014 inch gap was identified halfway around the

between the seat and the disc which was deemed acceptable for use

and the suction strainer leak was corrected.

On January 23, 1996, Unit 2 was synchronized on to the grid at

approximately 1630 hours MST.

ATTACHMENT 1 TO 9602280354 PAGE 1 OF 1

Arizona Public Service Company

PALO VERDE NUCLEAR GENERATING STATION

P.O. BOX 52034 o PHOENIX, ARIZONA 85072-2034

192-00960-JML/BAG/BE

JAMES M. LEVINE February 20, 1996

VICE PRESIDENT

NUCLEAR PRODUCTION

U. S. Nuclear Regulatory Commission

ATTN: Document Control Desk

Mail Station P1-37

Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)

Unit 2

Docket No. STN 50-529 (License No. NPF-51)

Licensee Event Report 96-001-00

Attached please find Licensee Event Report (LER) 96-001-00 prepared and submitted pursuant to 10CFR50.73. This LER reports a January 21, 1996, reactor trip on low water level following the degradation of main feedwater flow. The unit also received an Engineered Safety Feature Actuation System (ESFAS) actuation of the Auxiliary Feedwater Actuation System (AFAS) on low steam generator levels. In accordance with 10CFR50.73(d), a copy of this LER is being forwarded to the Regional Administrator, NRC Region IV.

If you have any questions, please contact Burton A. Grabo, Section Leader, Nuclear Regulatory Affairs, at (602) 393-6492.

Sincerely,

JML/BAG/BE/pv

Attachment

cc: L. J. Callan (all with attachment)

K. E. Perkins

K. E. Johnston

INPO Records Center

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